UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 9, 2009

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER NUCLEAR PLANT, UNIT NO. 3 – EVALUATION OF THE 2007 STEAM GENERATOR TUBE INSERVICE INSPECTIONS DURING REFUELING OUTAGE 15 (TAC NO. MD8918)

By letters dated May 30, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081550284), and November 18, 2008 (ADAMS Accession No. ML083370335), Florida Power Corporation (the licensee) submitted information summarizing the results of the 2007 steam generator tube inspections at Crystal River Unit 3 (CR-3). These inspections were performed during refueling outage 15. In addition to these reports, the U.S. Nuclear Regulatory Commission (NRC) staff summarized additional information concerning the 2007 steam generator tube inspections at CR-3 in a letter dated June 19, 2008 (ADAMS Accession No. ML081690349).

As discussed in the enclosed evaluation, the NRC staff has completed its review of these reports and concluded that the licensee provided the information required by CR-3 Technical Specifications and that no additional follow-up is required at this time. This completes the NRC staff's efforts under TAC No. MD8918.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure: As stated

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UNITED STATES
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EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

STEAM GENERATOR TUBE INSPECTION REPORTS FOR REFUELING OUTAGE

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

By letters dated May 30, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081550284) and November 18, 2008 (ADAMS Accession No. ML083370335), Florida Power Corporation (the licensee) submitted information summarizing the results of the 2007 steam generator (SG) tube inspections at Crystal River Unit 3 (CR-3). These inspections were performed during the refueling outage 15. In addition to these reports, the U.S. Nuclear Regulatory Commission (NRC) staff summarized additional information concerning the 2007 SG tube inspections at CR-3 in a letter dated June 19, 2008 (ADAMS Accession No. ML081690349).

CR-3 has two Babcock and Wilcox (B&W) once-through steam generators (OTSGs). Each SG contains 15,531 stress relieved mill annealed, Alloy 600 tubes. Each tube has a nominal outside diameter of 0.625 inches and a nominal wall thickness 0.034 inches. The tubes were mechanically roll expanded at both ends for approximately 1 inch of the 24-inch thick tubesheets and are supported by a number of carbon steel support plates. The hot-leg temperature is approximately 603 degrees Fahrenheit.

The NRC has approved a number of amendments related to the CR-3 SGs. Provisions for CR-3 SGs tube repair criteria are identified in paragraph "c" of improved technical specification (ITS) 5.6.2.10. "OTSG Program." The licensee may apply an intergranular attack indications depth criteria in the first span of SG-B or a tube end cracking repair criteria as an alternative to the 40 percent depth based criteria. The licensee is permitted to repair non-sleeved portion of the tubes by re-rolling them in the tubesheets or by sleeving in accordance with the provisions in ITS 5.6.2.10.f.

In May 30 and November 2008, letters, the licensee provided the scope, extent, method, and results of the CR-3 SG tube inspections. In addition, the licensee described corrective actions (i.e., tube plugging or repair) taken in response to the inspection findings. As a result of the review of the above inspection reports, the NRC staff has the following comments/observations:

1. Three tubes were in-situ pressure tested during the 2007 outage. Although all of the tubes passed the in-situ pressure test, there was a detectable change in the eddy current signal between the pre-in-situ and post-in-situ pressure testing inspections. The voltage and depth of the indications after the in-situ pressure tests were larger than the observed voltages and depths before the in-situ pressure test. All of the tubes that were pressure tested were stabilized and plugged.

Enclosure

2. The SGs at CR-3 will be replaced during the next refueling outage (currently scheduled for fall 2009).

3. The NRC staff review did not address the acceptability of the best-estimate, primary-to-secondary leakage expected during a large-break, loss-of-coolant accident (LBLOCA). This best-estimate determination was performed to satisfy a commitment made in support of a license amendment that permitted the use of a re-roll repair process for the CR-3 SGs.

The Pressurized Water Reactor Owner's Group (PWROG) is addressing the LBLOCA of concern on a generic basis in a topical report that is applicable to CR-3. The topical report was submitted on January 4, 2007 (ADAMS Accession No. ML070330123). The NRC staff believes that the generic PWROG program is the proper place to address the LBLOCA issue since the technical nature of this issue is complex, generic to B&W plants, and the licensee is planning to replace the CR-3 SGs during the next refueling outage.

Based on the review of the information provided, the NRC staff concludes that the licensee provided the information required by the technical specifications. In addition, the NRC staff concludes that there are no technical issues that warrant follow-up action at this time since the inspections appear to be consistent with the objective of detecting potential tube degradation and the inspection results appear to be consistent with industry operating experience at similarly designed and operated units. As discussed above, the NRC staff is reviewing the best-estimate, primary-to-secondary leakage expected for a LBLOCA with the PWROG on a generic basis for the B&W plants.

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Date: February 9, 2009

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Sincerely,
/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
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* **By a memorandum**

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